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Abstract

THE HETEROGENEOUS METHOD AND ITS APPLICATIONS

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Heterogeneous methods for lattice calculations are designed for use in cases where the lattice exhibits irregularities of a type that cannot be handled by the usual homogeneous methods. Such irregularities arise when control rods are inserted between fuel elements, in substitution experiments performed in exponential assemblies, or even in complete lattices containing zones of different fuel spacing as in the case of the Lucens reactor.

The usual two-group or age diffusion theory, in which fuel elements are represented by line sources and sinks, works reasonably well in lattices whose rods are thin compared to their spacing. The cluster elements proposed for D_2O moderated power reactors do not fall into this category, however, and introduce the additional complication of axial and radial streaming.

The heterogeneous method has therefore been extended by incorporating the following features:

(1) Three energy groups covering the fast region, the resonance region, and the thermal region. These groups are needed for two reasons. The first is to define the energy range in which resonance absorption takes place. The second is to introduce a buffer zone between the top group and the thermal group in order to produce a reasonably flat thermal neutron source.

(2) The flux at the fuel surface is expanded into a constant ("monopole")

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term plus a dipole term. The constant term describes the absorption characteristic of the element, the dipole term is required to describe properly the scattering properties of the fuel as well as radial streaming.

(3) Boundary conditions at the fuel surfaces are specified in terms of the linear extrapolation lengths into the fuel. The extrapolation length for the monopole term is well known. The extrapolation length for the dipole term is derived from calculated values of radial diffusion coefficients. Work is in progress to obtain them directly from transport theoretical arguments.

(4) The axial buckling inside a fuel element affects to some extent the linear extrapolation lengths, especially in the presence of voids. These effects are incorporated into the theory in order to account for axial streaming.

(5) The relatively large cluster elements require a more careful treatment of finite rod size than that of assuming line elements and later correcting for the finite rod radius. The present theory treats the finite radius rigorously, but breaks off the resulting series after a few terms.

The entire method is programmed in ALGOL and results will be presented.

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Abstract

NATURAL AND SLIGHTLY ENRICHED URANIUM LATTICE STUDIES OF VOIDED METAL ROD CLUSTERS IN D_2O

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Contributions by: B. Gloor**, W. Heer*, J. Marti**, E. Perpignani**, B. Rüegger*, E. Utzinger

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An extensive series of lattices consisting of voided rod cluster elements has been investigated in the subcritical facility MINOR. This experimental programme has formed the basis to check calculation methods for heavy water moderated and gas cooled pressure tube reactors. A detailed analysis was undertaken for an element consisting of 7 uranium metal rods enriched to 0.960% in U^{235}. This rod cluster is supported by a graphite structure which is surrounded by a pressure tube. An annular air gap separates the pressure from the calandria tube. This cluster presents a physical mockup of the fuel element to be used in the first charge of the Swiss experimental power station presently under construction at Lucens.

In order to obtain calculation methods which are of a more general usefulness and have a wider field of application the experiments specific to the Lucens reactor have been largely extended by varying a number of parameters of the lattice arrangements. The most important change consisted in a replacement of the enriched rods by natural uranium rods, leaving all structural material the same. The void content per cell was decreased by removing the calandria tube and increased by

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removing the graphite fillers. Finally some measurements with 4 rod clusters were performed. For all lattices several moderator to fuel volume ratios were investigated. For some or all of the lattices the following parameters were determined:

- The material buckling of single zone loadings by means of radial and axial flux plots (exponential). Some of the experiments were carried out in square as well as in triangular lattice geometry. Furthermore, the substitution technique was applied.
- Flux disadvantage factors of the different components of an element by means of flux fine structure measurements with Gold and Dysprosium foils.
- The epithermal component by means of the double foil technique.
- Cd-ratio of U^{238} by activation of depleted uranium foils.
- Axial fluxpeaking due to Al spacer pieces separating individual fuel slugs.

Theoretical interpretations of the experiments have been made on the basis of the two group four factor model. η is calculated according to the Westcott model. The fine structure of the thermal flux was obtained by P_3 theory adapted to the particular cluster geometry. The theoretical results are in agreement with the measurements for all the studied lattices within $\pm$ 2‰ in thermal utilization. Fast events were computed taking into account the heterogeneity of the element. Resonance escape probability p was obtained from a one group treatment assuming a constant slowing down density in the bulk moderator and calculating an extrapolation distance at the inner moderator surface with transport theory. Comparison with experiments indicates agreement within $\pm$ 1% in p. Diffusion coefficients have been calculated following Benoist's theory for the exact geometry by Monte Carlo techniques.

The measured bucklings have been compared with the calculations for all the investigated lattices. Agreement is within $20\text{-}30 \cdot 10^{-6}$ cm^{-2}.

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Abstract

The Lucens experimental nuclear power station

by

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Swiss industry, supported by most of the electric utilities of the country and by the Federal Government, is concentrating on the development of a single reactor type based on the pressure tube design with heavy water moderation.

The Lucens reactor is a first prototype along that line. It is cooled by carbon dioxyde. The fuel is metallic uranium, very slightly enriched due to the small size of the core, and canned with a magnesium alloy. In order to achieve a high power density, the fuel is subdivided in bundles of 7 rods per element. The mean coolant pressure is set at 60 at. Radial flattening of the neutron flux is achieved by dividing the core in two zones, the inner one having a smaller lattice-pitch. Control rods are driven in sequence, a design feature which will tend to flatten the axial burn-up distribution.

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Due to the thimble design of the pressure tubes, the hair pin coolant flow keeps the tube walls close to the cold leg temperature of the primary circuit. Furthermore it is possible to remove the pressure tubes easily with the fuel discharge-machine. This will be particularly useful for in-core testing of fuel elements of various designs.

Gas outlet temperature of the element is fixed at 380°C although the plant is designed for a higher temperature which may later be achieved by improved fuel elements.

The nominal thermal power of the reactor is 30 MW and the net electric power of the station is about 7 MW. A simple thermodynamic cycle has been selected for the plant without endeavouring to optimalize its efficiency.

The power station is built underground and is divided in three caverns, a reactor hall as containment with the reactor and the primary circuit, a fuel storage cavern and a turbine hall in which the turbine-generator set and its ancillary equipment are located. A service building houses the control room, electrical installations, offices, workshops and laboratories. In addition the site comprises a waste treatment plant, cooling towers, the river pumping station and a ventilation plant.

The total cost of the station is estimated at 70 million Swiss francs excluding heavy water and fuel. The station is presently under construction; start-up is scheduled for the end of 1965.

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Abstract

THERMAL PROBLEMS OF GAS-COOLED REACTORS FOR MAGNETOHYDRODYNAMIC POWER GENERATION

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Recent developments of magnetohydrodynamic (MHD) generators for large power production trend toward utilizing the possibility of inequilibrium non-thermal ionization for increasing the electron temperature over gas temperature thus enhancing the electrical conductivity of the flowing gaseous medium. Since the molecules of polyatomic gases possess internal degrees of freedom in addition to translational ones, they tend to absorb the excess energy of the electrons, thus only monoatomic gases in close circuit are suitable for electron temperature enhancement. A high temperature reactor, cooled by the monoatomic gas used as the flow medium for the MHD generator, is therefore the immediate choice.

Such a reactor presents stringent thermal design problems due to the requirement of intensive heat transfer and of moderate frictional power loss.

In this paper, these problems are discussed and methods of their solution given. It is shown that the ratio of frictional power loss to the heat rate and the ratio of the reactor core length to reactor core diameter are both related to the volumetric power density and to a linear dimension chosen as a characteristic length of the cooling matrix. A method is developed to take care of the large change of gas density during the passage through the reactor core. The difficulty of fulfilling the conditions imposed by the requirements of these quantities increases usually with the size of the reactor. The paper also shows that the requirements of the reactor and those of the MHD generator may dictate quite different choices of the most suitable gas medium.

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Abstract

GAS COOLING FOR FAST REACTORS

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Investigation of the possibilities of gas cooling for fast-breeder power reactors, including design work on a 1250 Mw(t) plutonium-fueled reactor of this type has been carried out at Zurich during the years 1962-1963. Interest in gas cooling for this purpose, apart from the desirability of having an alternate technology for fast reactors avoiding, as yet, unresolved practical engineering and operational difficulties of liquid metals, results from the possibilities for both improved fast reactor core neutronic performance and simplified engineering and operational features.

Core physics studies have shown coolant reactivity effects which are an order of magnitude smaller than for liquid metal cooled systems. This facilitates core designs with unity internal conversion ratio, while avoiding undesirably high positive coolant void and temperature coefficients and permits design for long intervals between refuelings.

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Fast spectrum calculations have been performed with the new GAM II code utilizing 99 fine group cross sections for a range of volumes and compositions. Other results of these calculations show a harder neutron spectrum for gas cooled core designs resulting in an increase of breeding ratio of 0.15 to 0.20 over that with sodium cooled cores.

The results of design studies are presented, including results which show that for cores in the 1,000 Mw(t) range, ratings in the region of 0.7 Mw/kg fissile, together with power densities of the order of 250 Kw/litre are readily obtainable, with helium or CO_2 cooling, at ambient pressures of 68 atmospheres for total pumping power expenditures of no more than 3 % of the thermal output. Investigation of alternate fuels for the system has been made with particular attention given to use of simple, round, stainless steel clad, oxide fuel elements with special design features to accommodate fission gas release for burnups in the 100,000 Mwd/t range. Operating conditions are substantially similar to present projected liquid metal-cooled designs and with maximum fuel element ratings in the 500 to 650 watt/cm range.

Surface roughening of the fuel element cladding is employed in a new, optimized way resulting in reducing the required number of fuel elements by a factor of four, while, at the same time, increasing the core specific power and power density. A new core thermal design code, GAZELLE III, which optimizes surface roughening, was developed to obtain this result.

The use of resonance absorbers, mixed with the core structural material to secure continuous decrease in reactivity with increase of water content, has been studied as a means of

dealing with steam-in-leak, and allowing use of water injection for emergency cooling. Detailed neutron spectrum calculations and reactivity determinations have been made for resonance absorbers including indium for a range of fast to soft spectrums. Appropriate resonance absorbers have small effect on the breeding ratio under operating conditions.

A layout has been made of a 500 Mw(e) power system incorporating a prestressed concrete pressure vessel enclosing the reactor core, series steam turbine driven circulators, and steam generators capable of supplying 565^{o}C steam. Special features are incorporated in the concrete vessel concept to require two independent failures before rapid loss of coolant. Over-all plant efficiency is about 38 %, taking into account coolant pumping requirements.

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Abstract

NON PERIODIC LATTICE WITH OPTIMUM K_{eff} VALUE AND NEGATIVE TEMPERATURE COEFFICIENT

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It is well known that the under-moderation necessary to obtain a negative temperature coefficient in a light water reactor is an economic waste. This inconvenience is remedied to a certain extent by the use of a multizone core. The authors propose to use a non-periodic lattice in which the uranium rods are disposed in concentric circles. The number of rods used in each circle is chosen so as to obtain a maximum K_{eff} value with a negative temperature coefficient, the total number of rods being constant.

The division into cylindrical cells caracterises this type of lattice in the following way : let circle "index k" have a radius R_k. The delimitation of cell index k will be known by the radii :

$$\frac{R_k + R_{k-1}}{2} \qquad \text{and} \qquad \frac{R_{k+1} + R_k}{2}$$

The coefficient of uranium concentration is :

$$g = \frac{\text{Uranium volume in one cell}}{\text{Total volume of the cell}}$$

and this coefficient caracterises each cell.

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The role of one cell can be compared to the role of one zone in a multizone reactor. To study the reactivity of a lattice as a function of the spatial distribution g (r) of the concentration coefficient, the authors use the theory of active and passive surfaces. This method, derived from the "Small Source Theory", considers the uranium rods as surface elements. The diffusion theory applies to the pure water placed between these surface elements, while the absorbing and fissile properties of uranium are taken into account by boundary conditions. The results of experiments with a subcritical assembly using slightly enriched uranium oxyde and light water confirm the theory. The temperature coefficient is determined by experiments at 20° C and 80° C. These measurements permit the establishment of the neutron balance and the deduction of the effective multiplication coefficient.

This study shows that it is possible to obtain a higher multiplication coefficient and a more uniform flux distribution in a non periodic lattice than in a regular under-moderated lattice, all at the same time conserving a negative temperature coefficient.

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Abstract

Special Problems concerning the Monotube Steam Generator in the Nuclear Power Station

by

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The paper deals with various problems resulting from the application of the monotube steam generator in the nuclear power station.

A brief summary of these problems is given in the following two main paragraphs:

Paragraph 1

We are dealing here in a general way with problems relating to computation, design, manufacture and erection as well as in regard to special operating conditions and questions of safety. With the aid of various examples of application in the field of gas-cooled reactors [CO_2 cooled reactors of the natural uranium type, advanced gas-cooled reactors (AGR), high temperature reactors (HTGR)] a few specific problems are taken and dealt with in the light of the experience obtained from conventional steam boiler construction as well as taking into account tests and studies which have been carried out in recent times. Special importance is attached to the so-called integrated steam generator, which is located in the same pressure vessel as the reactor. Isolated results of detailed project studies are referred to.

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Paragraph 2

This paragraph deals with the dynamic problems concerning the monotube steam generator. Basically, these can be divided into two groups, i.e., one group dealing with the major disturbances which occur under special operating conditions and the other group relating to small disturbances as occur during the control procedure under normal operation. By major disturbances we mean operating conditions that bring about a change in the thermodynamic factors within a wide range, for instance, starting up, shutting down, rapid closing down of the reactor, disturbances through failure of the control system etc. A simplified model for the non-linear simulation of the steam generator on the analogue computer is shown. With the aid of some arithmetical examples for the above-mentioned reactor types the problems are specifically dealt with.

The control procedure is treated by means of a linearized theory. In the first sub-section the possible control layouts for the various fields of application are mentioned and the state connections of the thermal equilibrium factors for various load conditions are explained with examples. The calculations for the temperature and pressure distribution, which in certain circumstances may be very extensive and time consuming, are solved on the analogue computer.

The second section refers to the transfer functions of the steam generator for the various possible disturbances. Furthermore, the behaviour of the entire plant is shown by calculation results. Here again endeavour is made to set up a comparison between the three above-mentioned main fields of application. In one example the agreement of the calculation with the results of the tests on a circuit is discussed.

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Abstract

Analogical study on the dynamic behaviour of the experimental nuclear power station at Lucens

by

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The behaviour of the Lucens station under variable conditions has been studied on a large-capacity analogue computer. The object of this work was to analyse the operation of the plant, to check its stability, to compare various modes of operation as well as different control systems, and finally to select the coefficients giving optimum performance.

In the first part the authors describe the control devices envisaged for the reactor, the primary loop, the steam generator and the turbo-alternator set. This equipment is devised so that the station can be controlled either from the reactor, to a predetermined power output programme (interconnected station), or from the distribution mains (isolated station). In the first case the nuclear output is fixed, whereas in the second case the reactor delivers the output demanded from it by the consumers. The circulation of coolant (CO_2) in the primary loop is always adapted to the nuclear output, so as to maintain a constant gas temperature at the reactor outlet.

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In a second part the bases of the calculation are set out. The capacity of the analogue computer, though considerable, was none the less limited, and it was necessary to adopt a certain number of simplifying assumptions. Only one steam generator is considered, instead of the two provided in the plant. The thermal models of the reactor and steam generator are subdivided into zones, for which the mean values are defined. The small time constants in relation to the behaviour of the whole plant are ignored.

The classical equations of neutron kinetics are introduced in their non-linear form, though only considering two imaginary groups of delayed neutrons. Other non-linearities are likewise preserved in the simulation of the reactor and of the primary loop. The models for the steam generator, the turbo-alternator set and the mains grid are described by linearized equations. The coefficients of the latter thus depend on the output level.

In the third part the authors show and comment on some examples of results obtained by simulation of the entire plant. The response of various characteristic values to small disturbances from a state of equilibrium at different output levels is recorded. With an interconnected station, for example, the disturbance consists of a variation in the set value of the power output, whereas for the isolated station a jump in the power absorbed by the mains is simulated. Various ways of regulating the steam generator are studied, also the individual influence of certain anticipatory signals.

It is confirmed that the station is capable of reacting rapidly to changes in the conditions demanded, and that the fluctuations of the controlled values remain within acceptable limits. In particular the gas temperature at the reactor outlet varies little, and the frequency variations observed in the isolated station are small.

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Abstract

HEAT TRANSFER INTENSIFICATION BY USE OF THE LONGITUDINALLY VARIABLE PRESSURE GRADIENT

by

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The problem of finding effective means of heat transfer intensification in gas-cooled reactors is still actual and progress in this direction may lead to significant improvements in reactor performances. The means used today, i.e. extended surfaces and surface roughening, are still not effective enough.

Especially high rates of heat transfer are found in fluids flowing normally to banks of closely spaced tubes and in beds of spherical particles. In both cases successive enlargements and narrowings in flow passages with large variations of the longitudinal pressure gradient are found. The study of fluid flow in diffusers reveals longitudinal vortices formed on the walls which break up on the edge of the boundary layer thus causing a high degree of turbulence in the stream. This effect must have an influence on the heat transfer rates. On the other hand, it has been proved that the degree of turbulence influences heat transfer in converging passages.

On the ground of the above analysis a preliminary investigation has been undertaker of heat transfer in an annular diffuser which has shown a substantial increase in heat transfer rates in comparison with the flow with a zero pressure gradient. The experimental investigation proceeded in a two-dimensional apparatus described in this paper. The test section consisted of a rectangular channel with two walls profiled in such a way as to form a succession of diffusing and converging sections with a half angle of 8°. The profiled walls were heated electrically and local wall temperatures measured along the channel by numerous thermocouples. The velocity and temperature profiles in the fluid were measured in four cross-sections. The static pressure distribution along the test section was also measured.

The experimental data show a considerable influence of the adverse pressure gradient on the heat transfer rate. The Stanton number based on the local flow area increases steadily along the diffuser up to more than twice in comparison with the flow without a pressure gradient. The intense turbulence created in the diffusing section influences substantially the heat transfer rate in the converging section too. The pressure loss coefficient which increases along the diffuser is greatly lowered in the converging section indicating a pressure energy recovery in this section.

The above results, obtained for a single geometry, show that the pressure gradient effect can be used successfully for heat transfer intensification with gas cooling. Other geometries are being tested in order to find an optimum combination of parameters which influence heat transfer and fluid flow.

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Abstract

BOILING FROM A LIQUID INTERFACE

by

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The paper presents the results of an investigation of boiling heat transfer from an ideally smooth surface, liquid to liquid interface. In this way, surface roughness is eliminated.

An original apparatus is described which eliminates by-effects (e.g. boiling from solid surface). Horizontal mercury surface was used as the heating surface from which four liquids were boiled at atmospheric pressure. The liquid surface (interface) was kept very clean by a special experimental procedure. The boiling liquids were water, ethyl alcohol, benzol and n-pentane, all with boiling points from 35 to 100°C.

The results of measurement of the boiling heat flux q, superheating of the heating liquid surface ΔT and the corresponding heat transfer coefficient α are presented. The superheating of the heating (liquid) surface was found to be higher than that of boiling from solid surfaces.

Two boiling regions, visually similar, are clearly distinguished, convective boiling and nucleate boiling; conditions of transfer from one kind of boiling to the other are shown. The heat transfer coefficient as a function of the driving temperature difference (surface superheating) for both regions can be represented by

$$\alpha = C(\Delta T)^n$$

where C is a constant dependent on physical parameters of boiling liquids and n a constant for all boiling liquids in the nucleate boiling region.

The heat transfer results for all liquids are correlated to the physical properties of boiling liquids and represented in the dimensionless form

$$Nu = f(Pr, Pe, Kp)$$

where variables Nu, Pr, Pe, and Kp are dimensionless numbers based on bubble parameters.

A preliminary study of nucleation mechanism by fast cine camera (6000 frames/sec) is reported. It was found that the bubble frequency increases with the increasing departing bubble diameter, which is opposite to the experimental results for boiling from solid surface. The possibilities of further detailed fast cine studies are discussed.

In conclusion, possibilities of practical application of this phenomenon for high efficiency-high flux heat transfer and heat generating apparatuses, including power reactors, are discussed.

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Abstract

SINTERED FUEL OF HIGH DENSITY OBTAINED BY "METALLIC GRADE" UO_2 POWDER CHARACTERIZED BY SEDIMENTATION ANALYSIS REGARDLESS OF ORTHOKINETIC FLOCCULATION

by

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Ceramic fuel containing sintered uranium dioxide is normally obtained from "ceramic grade" UO_2 powder. Our trends have been directed towards obtaining this fuel from the "metallic grade" UO_2 powder, although this method is unusual. The characteristics of uranium dioxide powder have been therefore determined by sedimentation analysis and at the same time the influence of flocculation on the values of characteristic parameters of the powder observed. It has been attempted to define a parameter characterizing the powder which is independent of experimental conditions. In accordance with our earlier theoretical and experimental work it has been shown that this characteristic value can be also found in this particular case. The weight mean diameter of uranium dioxide is linearly dependent on concentration and can be extrapolated to zero from which the characteristic parameter of the powder is defined.

An analysis of the experimental results also indicates that orthokinetic flocculation of the uranium dioxide investigated is not particularly pronounced in some metal powders.

Investigations of the "metallic grade" uranium dioxide powder by the differential thermal analysis method have shown that its fraction below 60 microns has a characteristic temperature difference of 158°. This indicates that it belongs to the group of sinterable powders which have been proved by simultaneous investigations of the sintering of "metallic grade" uranium dioxide powder prior to fractionation and of the separated fraction.

By sintering the "metallic grade" uranium dioxide powder whose agglomerates are below 60 microns in an atmosphere of water vapor or carbon dioxide at 1300° densities up to 96% of theoretical values can be obtained. This indicates that high density sintered fuels can be easily obtained from "metallic grade" uranium. The characteristic fracture through grains and the adequate grain size have been observed by fracture surface and microscopic studies.

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Abstract

METALLURGICAL PROPERTIES OF SOME URANIUM ALLOYS

by

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Fxperimental work on some dilute uranium alloys, especially on the uranium-molibdenum system, is reported.

An alloy with 0.5 wt. % Mo (total impurities in uranium about 1000 ppm) in as-cast condition has fine fairly equiaxed grains ranging from 150 to 250 microns. The grains are completely occupied by a lamellar perlite. This alloy was subjected to long term annealin in a high alpha phase temperature range. It was fairly convincingly shown that the grains did not undergo any pronounced enlargement by annealing at 450 to 550°C for 2500 hours. In the alloy subjected to annealing at 550°C for more than 2500 hrs., an observable grain growth took place. Isothermal treatment of the alloy in the range of the upper "C" curve yields, after the transformation is complete, relatively fine alpha grains whose size decreases with decreasing temperature of the treatment. The average grain size obtained by isothermal treatment at 550°C was 35 microns and did not significantly change after heating for 2000 hrs. at 550°C. Quenching from the gamma range in water and salted water gives, besides the beta phase, a fine needlelike α'_a phase. This martensitic phase is athermal and probably results from the direct decomposition $\gamma \rightarrow \alpha'_a$. Quenching from the $\beta+\gamma$ and β range gives no athermal martensitic phase.

The ternary system with aluminum shows a pronounced cooperative effect of molibdenum and aluminum upon the grain size in the as-cast alloy. The finest and most regular grains (50 microns in average) are obtained with the 0.6 wt. % Mo and 500 ppm Al alloy. The modified Jominy-test shows that the cooling rate has a less effect upon the alpha grain size in this alloy than in the U-O.5 wt. % Mo alloy.

The present investigation of the kinetics of phase transformations in the U-Nb system (up to 5 wt. % Nb) indicates a slight beta phase stabilization. In the range of 2-3 wt.% Nb a significant change in microstructure and lattice parameters of alpha uranium were observed.

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Abstract

ON THE URANIUM-HYDRAZINE SYSTEM

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Several reactions of the non-aqueous and aqueous hydrazine with different uranium salts (UF_6, UF_4, $UO_2(NO_3)_2$ etc.) are described which could be possibly used as intermediate steps in the preparation and purification of uranium fuels.

The products of the reaction of non-aqueous hydrazine with uranium fluorides (UF_6 and UF_4) were analysed and compared with compounds obtained by the reaction of hydrazine fluoride with metallic uranium and uranium oxides.

In the system hydrazine hydrate-uranyl nitrate-water different precipitates with uranium in (+6) and (+4) valence states were obtained depending on the reaction conditions. The precipitates were decomposed and several uranium oxides were found to be formed during the decomposition process.

The basic apparatus for the experimental work under specified conditions is also described.

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Abstract

URANIUM DIOXIDE PRODUCTION. A SURVEY OF ELECTROCHEMICAL REDUCTION AND PRECIPITATION FROM CARBONATE SOLUTIONS

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Conditions for the electrochemical reduction of uranium(VI) in carbonate solutions to uranium(IV), and conditions for the subsequent precipitation of uranium dioxide (or hydrated oxides) are given and the obtained results discussed.

The electrochemical reduction of uranium(VI) in carbonate solutions proceeds by one-electron-step reaction to uranium(V) carbonate complex, which subsequently dismutates to uranium(VI) and uranium(IV) carbonate complexes by a second order kinetics. The dismutation rate depends on pH and on the total carbonate concentration. The constant potential electrochemical reduction on the mercury electrode seems to be the only possible way to obtain stable uranium(IV) carbonate complex solutions. The reduction of uranium(VI) on platinum and nickel electrodes was not selective. Therefore, these metals could not be used as electrode materials in the reduction process.

The conditions for the precipitation of uranium(IV) from stable solutions are described. The precipitation is induced either by raising the temperature to 60°C, or by raising the pH of the solution, or by dilution. Concentration limits of the precipitation are shown by plotting the concentration of total sodium carbonate vs. concentration of uranium.

The reduction of uranium(VI) on mercury cathodes of different areas (1 : 10 : 100) was carried out. The current density on the cathode depends on the rate of stirring at given experimental

conditions. In order to obtain the optimal current densities a pilot-plant electrolysis cell with a rotating mercury electrode was constructed. The cell had an operating volume of 200 liters. The cathode potential was electronically controlled.

On the basis of the results obtained, the flow sheet of a cyclic process is proposed. Two aspects of application are discussed.

1. Production of uranium dioxide from ammonium diuranate by electrolysis of ammonium carbonate solutions of high uranium content;
2. Extraction of uranium from ore leach liquors to obtain technical concentrates. Application to low grade ore leach solutions is considered also.

The precipitate, obtained from solutions of high uranium concentrations, shows regularly an O/U ratio lower than 2.10. After drying at 300°C, in inert atmosphere, the products had cubic crystal structures determined by X-ray analysis.

The pellet of uranium dioxide pressed at 4000 kg/cm^2 and sintered at 1350°C had densities up to 10.2 g/cm^3.

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Abstract

PRELIMINARY STUDY OF A FUEL REPROCESSING PILOT PLANT

by

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Recently a group of engineers from Noratom A.S., Boris Kidrič Institute of Nuclear Sciences, and Institutt for Atomenergi have made in co-operation a preliminary design study of a fuel-reprocessing pilot plant. The present paper is a short description of the work performed.

The purpose of the reprocessing facility is: (1) to reprocess fuel of some experimental reactors, (2) to develop fuel reprocessing of a future power reactor, (3) to gain experience and technical "know-how" in the field of fuel reprocessing.

The pilot plant is primarily designed for aqueous reprocessing of natural and/or low-enriched metallic uranium, canned in pure or alloyed aluminium or magnesium. However, oxide fuels in stainless steel or Zircalloy could also be treated by a "chopping and leaching" dissolution technique. The anticipated fuel burn-up is 3000-7000 MWD/t, and the cooling-time about 120 days. The nominal plant throughput is 1 kg U per hour, but if necessary, it can be increased.

The basic chemical flowsheet is an adopted Purex process consisting of chemical (or mechanical) decanning and dissolution of fuel, two extraction cycles for uranium and plutonium, "tail-end" uranium purification and auxiliary processes (i.e. waste treatment, acid and solvent recovery). In the extraction flowsheet, backcycling of the second cycle waste solutions is provided. Further, to increase the decontamination and concentration

of the plutonium product, a reflux flowsheet has been selected for the second plutonium cycle, or, as an alternative, an amine extraction of plutonium.

The chemical flowsheets developed in the present study are used as a basis for the plant design. However, the design is made sufficiently flexible to permit process variations within wide limits.

The main process equipment includes one dissolver, nine extraction columns, four evaporators, several towers (for acid absorption and rectification, steam-stripping), equipment for uranium "tail-end" treatment and solvent recovery. This equipment is contained in six shielded cells located in the central part of the main building. The main building also includes a "hot" and a "cold" wing, with auxiliary facilities. In addition to the main building, two small buildings are provided, one for waste storage and the other for transformers and compressors.

An economic evaluation which includes the estimated investment cost and the direct annual operating cost is presented.

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Abstract

AN ANALYTICAL METHOD FOR NEUTRON THERMALIZATION CALCULATIONS IN HETEROGENEOUS REACTORS

by

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It is well known that the use of diffusion approximation for studying neutron thermalization in heterogeneous reactors may lead to considerable errors. On the other hand, exact non-diffusional methods, which are numerical and rather laborious, require the use of large digital computers. In this paper the use of diffusion approximation in absorbing media has been avoided but the treatment remained analytical. This made possible a clearer insight into the physics of the phenomenon and a simpler way of practical calculations.

The moderator is treated according to the heavy gas model, the slowing down in the fuel is neglected, and the angular distribution of neutrons at the fuel boundary is assumed to be isotropic. The non-Maxwellian factor of distribution in the moderator is expressed as a sum of an exact one-velocity solution and a series of Laguerre polynomials multiplied by Bessel functions. An energy dependent boundary condition on the fuel surface has been determined by the neutron balance method without the use of diffusion approximation and the resulting formula has been connected with corresponding escape probabilities whose energy dependence is determined from basic cross-section data. The effects of using different escape probabilities have been studied. Explicit formulae for calculating disadvantage factors of the fuel and the moderator, the mean neutron energy and the neutron temperature, have been derived. The convergency of the expansions has shown to be satisfactory. The method has been adapted for practical calculations with different temperatures and different cylindrical fuel elements including claddings.

To check the method, heavy water and graphite cells were calculated by using both the diffusional and the more exact methods with and without taking thermalization into account. The results of the proposed method have shown to be considerably more accurate than the diffusional ones and they are in good agreement with Honeck's numerical and experimental results. Finally, the method has been used for theoretical analysis of experiments on the Yugoslav RB reactor.

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ONE-VELOCITY NEUTRON DIFFUSION CALCULATIONS BASED ON A TWO-GROUP REACTOR MODEL

by

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In evaluating critical dimensions and flux distributions in multiregion nuclear reactors it is advantageous to have a method that retains computational simplicity of one-group treatment and yet approaches the accuracy attainable with a two-group model. Such analytically simple but sufficiently accurate representations of the flux equations are particularly important in dynamic studies when it is necessary to obtain simultaneously the solution of the flux equation and its adjoint on an analog computer.

The paper describes a procedure by which the flux distribution can be represented with an expression of the form:

$$F^i(r) = F_1^i(r) + Z_i^i F_1^i(R_i) \exp[-V_i^i(R_i - r)] + Z_{i-1}^i F_1^i(R_{i-1}) \exp[V_{i-1}^i(R_{i-1} - r)]$$

where $F_1^i(r)$ is the solution of the one-group diffusion equation in the i-th region of the reactor, defined by the interval: $R_{i-1} \leqslant r < R_i$, while Z_i^i, Z_{i-1}^i, V_i^i, and V_{i-1}^i are constants.

At each interface between two regions, the function $F_1^i(r)$ must satisfy a boundary condition defined by:

$$\lim_{r \to R_i} \frac{dF_1^i(r)}{dr} \cdot \frac{1}{F_1^i(r)} = U_i^i$$

In deriving this expression a one dimensional two-group reactor model, adapted for solution on an analog computer, is used. The two-group diffusion equations are transformed into Riccati equations with associated initial and boundary conditions. Using the properties exhibited by the solution of Riccati equations, the multiregion reactor is reduced to only two regions, involving a simple set of equations which determine the critical dimensions and the coefficients:

$$Z_i^i,\ Z_{i-1}^i,\ V_i^i,\ V_{i-1}^i,\ U_i^i$$

for all regions.

The normalized set of these equations can be solved once for all, whereby a family of curves is obtained, from which critical dimensions and coefficients can easily be determined for various reactor parameters.

This procedure is applicable to both spherical and cylindrical geometry and to any arrangement of active zones and reflectors. The practical application and accuracy of this procedure is discussed in specific examples.

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Abstract

AN ANALYSIS OF CRITICAL EXPERIMENTS USING THE HETEROGENEOUS REACTOR THEORY

by

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The heterogeneous reactor theory is used for the analysis of critical experiments performed on one and multizone reactor cores. The main reason for this use was to overcome the difficulties which arise in the interpretation of critical experiments when relatively a small number of fuel elements is present and classical flux mapping techniques do not give satisfactory results.

The heterogeneous reactor theory in the two-group or continuous slowing down model offers the possibility of determining two main parameters of fuel elements: η, the number of neutrons emitted per neutron absorbed in the fuel, and γ, the number of neutrons absorbed per unit neutron flux on the fuel surface. For this purpose the number of fuel elements and core height in critical conditions have to be experimentally determined. From the results of two experiments with a different number of fuel elements independent quantities η and γ can be determined. In two-zone experiments with known fuel parameters in the reference zone and unknown in the test zone, two successive substitutions allow the determination of unknown fuel parameters.

This can be extended to the three-group model which makes possible independent determination of resonance absorption parameters of the fuel. In this case it is assumed that resonance absorption takes place on a single energy line between the first and second neutron energy group. For experimental determination of the resonance parameter it is necessary to undertake three independent critical experiments with a different number of fuel elements, or, in a two-zone system, to procede with three successive substitutions.

A series of one- and two-zone experiments performed on the RB zero power reactor using 2% enriched and natural uranium in different lattice configurations were analysed by the described method. The obtained results were tested for consistency and reproducibility and compared with the results of the standard interpretation of critical experiments based on homogenization principles. The advantages of the heterogeneous method are evident especially in two-zone substitution experiments. With a sufficient number of fuel elements both methods give equivalent results.

In conclusion, advantages and disadvantages of the heterogeneous interpretation of critical experiments are discussed.

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LEAKAGE SPECTRUM OF THERMAL NEUTRONS

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A short review of the variational-iterational method of solving the integral transport equation for thermal neutrons in a semi-infinite source-free absorbing medium is given. This method is used to obtain explicit expressions for the linear extrapolation distance and leakage flux of thermal neutrons in terms of the corresponding infinite medium solution.

For illustration, a classical monatomic gaseous moderator has been chosen for which the infinite medium solution has already been obtained semi-analytically. The absorption cross-section is supposed to follow the 1/v law. The numerical part of the problem has been solved on the IBM 704 digital computer.

Numerical results for the linear extrapolation distance for different atomic mass numbers with several degrees of the absorption probabilities of the medium are presented. Also, the effective temperature of leaking neutrons is discussed as a function of the atomic mass, the absorption of the medium, and for different directions of neutron leakage.

Theoretical results are discussed in comparison with the experimental values.